October 7, 1999

Mr. Robert Wanczyk
Acting Director of Operations
Vermont Yankee Nuclear Power Corporation
185 Old Ferry Road
Brattleboro, Vermont 05301

SUBJECT: NRC INSPECTION REPORT 50-271/99-11

Dear Mr. Wanczyk:

On August 27, 1999, the NRC completed an inspection at the Vermont Yankee (VY) Nuclear Power Station. The inspection focused on the effectiveness of the engineering functions in providing for the safe operation of the plant. The enclosed report presents the results of that inspection. The results were discussed with you, Mr. J. Hoffman and other members of your staff on August 27, 1999.

Overall, the team concluded that the engineering organization provided acceptable support to plant operations. We noted that the design change process procedures were appropriate, and that the quality of design changes, associated safety evaluations, and calculations was acceptable. We found engineering provided appropriate support in identifying corrective actions for plant events and that operability determinations were reasonable. We noted that the design bases reconstitution effort and the Final Safety Analysis Report Verification Program were effective and comprehensive. In addition, we determined your self-assessment program was effective in identifying areas for improvement.

The NRC has determined that a violation of NRC requirements occurred regarding the lack of cable separation in the upgrade modification of the residual heat removal (RHR) and core spray (CS) monitoring instruments to the RG 1.97 Category 1 criteria. This violation is being treated as a Non-Cited Violation (NCV), consistent with Appendix C of the Enforcement Policy. This NCV is described in the subject inspection report. You do not need to respond to the violation, but you may contest the violation or severity level. If you so choose, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Vermont Yankee facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room (PDR).

We appreciate your cooperation.

Sincerely,

ORIGINAL SIGNED BY:

Lawrence T. Doerflein, Chief
Engineering Programs Branch
Division of Reactor Safety

Docket No.: 50-271
License No.: DPR-28

Enclosure: NRC Inspection Report 50-271/99-11

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U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.	50-271
Licensee No.	DPR-28
Report No.	99-11
Licensee:	Vermont Yankee Nuclear Power Corporation
Facility:	Vermont Yankee Nuclear Power Station
Location:	Vernon, Vermont
Dates:	August 9, 1999 - August 27, 1999
Inspectors:	Aniello L. Della Greca, Senior Reactor Engineer Barry S. Norris, Senior Reactor Engineer Keith A. Young, Reactor Engineer Gregory V. Cranston, Reactor Engineer
Approved by:	Lawrence T. Doerflein, Chief Engineering Programs Branch Division of Reactor Safety

EXECUTIVE SUMMARY

Three Mile Island Nuclear Power Station
Report No. 50-289/99-05

During the period between August 9 and August 27, 1999, the NRC conducted an engineering team inspection at the Vermont Yankee Nuclear Station. The objective of the inspection was to assess the effectiveness of the engineering functions in providing for the safe operation of the plant. The results of this inspection are described below:

Maintenance

! The licensee's operability determination following a temporary loss of the standby liquid control (SLC) "B" sub-system explosive valve monitoring circuit was acceptable. However, the inspectors noted the licensee does not routinely declare Technical Specification equipment inoperable during surveillance activities. This issue, which the licensee previously identified and is evaluating, will be a NRC follow-up item. [M1.1]

Engineering

! The design changes, both temporary and permanent, were properly documented and evaluated in accordance with 10 CFR 50.59 to ensure that they did not involve an unreviewed safety question. Design drawings and calculations had been properly revised and developed, as applicable. No concerns were identified with the installation or post-modification testing of the revised design. [E1.1]

! The engineering calculations reviewed were acceptable, adequately controlled and in accordance with the existing procedure. The assumptions, methodology, and conclusions were typically appropriate and acceptable, although occasionally some of the assumptions or their basis were not clearly stated. [E1.2]

! The system design basis documents were consistent with the Final Safety Analysis Report (FSAR), comprehensive, appropriately updated and controlled, and had clear reference to supporting documentation. [E1.3]

! The FSAR Accuracy Verification Program was effective in finding and correcting the FSAR discrepancies. Discrepancies were appropriately reviewed and dispositioned in accordance the program procedures and/or the corrective action program. The FSAR revisions were properly implemented. [E1.4]

! The engineering response to Event Reports in support of plant operations was acceptable with appropriate bases for operability assessments and reportability determinations. The problems and corrective actions identified were appropriately entered into the corrective action program for tracking and closure. [E2.1]

Executive Summary (cont'd)

- The licensee made significant improvement in addressing degraded conditions and reducing the number of open Basis for Maintaining Operation (BMO) determinations. Also, the process for operability determinations (BMOs) was clarified. [E2.2]

- The licensee was acceptably managing the engineering work backlog and was making good progress in developing trend data. [E2.3]

- The procedures for the design, evaluation, installation and testing of permanent and temporary plant modifications were acceptable and sufficiently detailed to ensure proper implementation of the change process and to ensure that the change did involve an unreviewed safety question. [E3.1]

- The system engineering organization was relatively new and included several engineers that had either limited system experience or were new to the site. Therefore, the effectiveness of the organization appeared limited. Nonetheless, the engineers interviewed indicated a strong desire to take on the assigned responsibilities and be directly involved in the activities related to their respective systems. [E5.1]

- The engineering program evaluations conducted by the Quality Assurance (QA) organization and self-assessments conducted by Engineering were effective in identifying engineering program areas for improvement. The QA audits were broad in scope and addressed most engineering programs. Findings were formally tracked and observations appropriately reviewed and resolved. [E7.1]

- No analysis was available to address the impact on the operability of some valves if the movement of the thin rod attached to the valve stem for local position indication was obstructed in some manner. This item will be reviewed during a future inspection. [E8.1]

- The licensee’s review and actions to resolve the cable separation issue created during the upgrade of the residual heat removal (RHR) and core spray (CS) instruments to the Regulatory Guide 1.97 Category 1 criteria were acceptable. However, this non-repetitive licensee-identified and already corrected violation of 10 CFR 50, Appendix B, Criterion III requirements is being treated as a Non-Cited Violation, consistent with Appendix C of the NRC Enforcement Policy. [E8.4]

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Summary of Plant Status

During the period between August 9 and August 27, 1999, the NRC conducted an engineering team inspection at the Vermont Yankee Station. The objective of the inspection was to assess the effectiveness of the engineering functions in providing for the safe operation of the plant. The plant remained at or near full power throughout the inspection period.

II. MAINTENANCE

M1 Conduct of Maintenance

M1.1 Surveillance Conducted Without Entering a Limited Condition for Operation (LCO)

a. Inspection Scope (37550)

The team evaluated the operability of the standby liquid control (SLC) system due to the explosive valve monitoring circuit being disabled during the associated continuity meter calibration. The evaluation included discussions with station management.

b. Observations and Findings

Vermont Yankee (VY) Event Report (ER) #99-878 noted that the fuse for the “B” SLC system explosive valve continuity meter blew following the meter calibration. The ER also noted that a replacement fuse was not readily available due to a receipt inspection hold. The VY Final Safety Analysis Report (FSAR), Section 3.8.3, states that the continuity of the discharge valve ignition circuit is monitored by a trickle current, an alarm, and indicator lights. Similarly, the SLC Design Basis Document (DBD), Section 3.2.4, states that the ignition circuit of each explosive actuated valve shall be continuously monitored for continuity and short circuits.

Since the continuity circuit was disabled until a fuse could be obtained, the team questioned the basis for considering the “B” SLC sub-system operable during that period. The team learned that the operability determination was based on the continuity circuit not affecting the ability of the system to perform its required safety function. The team reviewed the SLC system monitoring circuit and considered the licensee’s basis acceptable.

During their review of this issue, the team also determined that VY did not usually declare Technical Specification equipment inoperable during routine surveillance activities. Operations management had recently identified this program discrepancy and were developing a plan to resolve the issue. The NRC will review the actions to address this licensee-recognized discrepancy at a later date. **(IFI 50-271/99-11-01)**

c. Conclusion

The licensee's operability determination following a temporary loss of the standby liquid control (SLC) "B" sub-system explosive valve monitoring circuit was acceptable. However, the inspectors noted the licensee does not routinely declare Technical Specification equipment inoperable during surveillance activities. This issue, which the licensee previously identified and is evaluating, will be a NRC follow-up item.

III. Engineering

E1 Conduct of Engineering

E1.1 Safety-Significant Design Modifications

a. Inspection Scope (37550, 37001)

The team reviewed permanent and temporary modification packages to assess the quality of engineering analyses and verify that the design change process complied with plant procedures and NRC regulatory requirements. The team also reviewed the associated safety evaluations to determine if unreviewed safety questions existed. Additionally, the team conducted walkdowns of selected plant modifications to confirm that the installations conformed with design documents.

b. Observations and Findings

The team selected ten safety-significant permanent modifications and ten temporary modifications for their review. Most permanent and temporary modifications were recently installed in the plant. One permanent modification had been designed and scheduled to be implemented during the fall 1999 refueling outage.

The team's review of permanent modification packages found that the purpose of the modifications was clearly defined and that the design changes were detailed and appropriate for the intended purpose. The team also found that the documents requiring updates had been revised and that the applicable calculations had been either updated or prepared, as needed. The 10 CFR 50.59 safety evaluations were detailed and correct in their determination that the changes did not constitute unreviewed safety questions. Additionally, appropriate post-modification testing had been accomplished prior to release of the changes to operations. Walkdown of two modifications revealed no discrepancies between the as-installed configurations and the modification drawings. The modification packages and safety evaluations met their respective procedural requirements.

The team's review of the temporary modifications packages found that the design changes were technically acceptable and that they had been properly evaluated by the licensee to ensure that the design bases of the plant had been maintained. The team also found that some temporary modifications had been in existence for more than two years. The licensee had recognized the need to better control the temporary modification process and had revised the applicable procedure AP 0020, "Control of Temporary and Minor Modifications," to require that temporary modifications be evaluated on a quarterly basis and justifications be presented to the Plant Operations Review Committee (PORC) if they are required beyond a six-month period.

c. Conclusions

The team concluded that the Vermont Yankee design change process was acceptably implemented. The design changes, both temporary and permanent, were properly documented and evaluated in accordance with 10 CFR 50.59 to ensure that they did not involve an unreviewed safety question. Design drawings and calculations had been properly revised and developed, as applicable. No concerns were identified with the installation or post-modification testing of the revised design.

E1.2 Engineering Calculations

a. Inspection Scope (37550)

The team reviewed selected calculations prepared by design engineering to evaluate their scope and content and to determine acceptability of the assumptions, methodology, and conclusions.

b. Observations and Findings

The team found that the scope and content of the calculations were consistent with their intended purpose and that sufficient details had been provided by the licensee regarding the methods used and the intended results. The assumptions were reasonable and, in general, the bases for assumptions were provided, although occasionally some of the assumptions were not clearly stated. Also, in one case, the team found that the basis for a key assumption in one of the calculations was not provided. Upon additional discussions with the licensee, the team determined that an adequate basis existed which the licensee had not documented in the calculation. The licensee acknowledged that the basis should have been documented.

The methodology and calculation methods used were appropriate for the purpose of the calculation and supported the conclusions reached in the calculation regarding system and component operability. The calculations were detailed and had been appropriately reviewed.

The team's review of selected portions of engineering procedure WE-103, "Engineering Calculations and Analyses," noted that the procedure provided sufficient guidance for the proper control of the calculations and for the correct processing of the calculation results. No concerns were identified with the processing of the calculations and the procedure implementation.

Recently the licensee initiated a program to improve their control of calculations. The program involved the review of existing calculations for status, applicability, and cross referencing to other calculations. The review also included the update of the Calculation Index and the identification of superceded calculations.

c. Conclusion

The engineering calculations reviewed were acceptable, adequately controlled and in accordance with the existing procedure. The assumptions, methodology, and conclusions were typically appropriate and acceptable, although occasionally some of the assumptions or their basis were not clearly stated.

E1.3 Control of Design Bases

a. Inspection Scope (37550).

The team reviewed how engineering ensured preservation of the plant design bases and how design modifications were factored into the systems design basis documents.

b. Observations and Findings

The team's review of the VY design bases reconstitution effort determined that the licensee had completed 23 Design Basis Documents (DBDs) with an additional six in progress. Of the incomplete DBDs, five were topical design basis documents covering seismic, event combinations, functional requirements, external events and internal events. The team selected the DBDs for the Residual Heat Removal and Residual Heat Removal Service Water systems for review. The team found the DBDs to be comprehensive in content, appropriately updated and controlled, and with clear reference to the supporting documentation. Also, the DBDs were consistent with the FSAR.

The team's review of selected plant modifications found that the design changes had been correctly posted against or incorporated into the design and licensing basis documents such that the plant as-built status was consistent with the design drawings and the FSAR.

c. Conclusion

The system design basis documents were consistent with the Final Safety Analysis Report, comprehensive, appropriately updated and controlled, and had clear reference to supporting documentation.

E1.4 Engineering FSAR Accuracy Verification Program

a. Inspection Scope (37550)

The team reviewed the licensee's Final Safety Analysis Report (FSAR) Accuracy Verification Program to confirm that the Program was effective in identifying and correcting FSAR errors and inconsistencies.

b. Observations and Findings

The team's review of the FSAR Accuracy Verification Program determined that the effort involved the review of FSAR statements against other sections of the FSAR and other licensing and design bases documents, including the plant technical specifications, the plant design drawings, the engineering and setpoint calculations, engineering memoranda, and the input assumption source documents. Based on a sample review of condition reports, the team determined that the project was effective in identifying discrepancies. The identified discrepancies were properly tracked, and each finding was reviewed and dispositioned in accordance the program procedures and/or the VY corrective action program. FSAR revisions were accomplished in accordance with procedure AP-6036, "FSAR Revision Process."

c. Conclusion

The Final Safety Analysis Report (FSAR) Accuracy Verification Program was effective in finding and correcting FSAR discrepancies. Discrepancies were appropriately reviewed and dispositioned in accordance the program procedures and/or the corrective action program. The FSAR revisions were properly implemented.

E2 Engineering Support of Facilities and Equipment

E2.1 Engineering Support of Site Activities

a. Inspection Scope (37550)

The team reviewed selected Event Reports (ERs) that were addressed by Engineering to determine if the actions taken in support of the plant event were appropriate, the problems were properly reviewed for extent of condition, the causes were correctly identified, and the bases for operability and reportability determinations, when applicable, were adequately supported.

b. Observations and Findings

The team observed that the ERs reviewed comprehensively addressed the issue in the problem statement and that, in dispositioning the ERs, Engineering appropriately identified both the immediate and the long term corrective actions. The operability assessments, the bases for maintaining operation, and the reportability determinations had been performed, when required, to adequately support the licensee's position regarding system operability. The causal determinations provided adequate information for corrective actions and/or trending. The problems and corrective actions identified were appropriately entered into the corrective action program for tracking to closure.

c. Conclusion

The engineering response to Event Reports in support of plant operations was acceptable with appropriate bases for operability assessments and reportability determinations. The problems and corrective actions identified were appropriately entered into the corrective action program for tracking and closure.

E2.2 Operability Determinations

a. Inspection Scope

The team reviewed the process for the generating operability determinations. The review evaluated selected portions of the plant procedures, the status reports, and the open operability determinations. The team also conducted interviews with system engineers, and engineering and operations management.

b. Observations and Findings

At VY, operability determinations were part of a process called "Basis for Maintaining Operability" (BMO). Previously, the process was contained in a guideline that supported the corrective action program. The process was formalized into a procedure that was issued on July 26, 1999 (AP 0167, "Operability Determinations"). The procedure did not significantly change how BMOs were performed, but provided additional guidance and clarifications.

Licensee management reviewed all open BMOs on a weekly basis as part of the on-site review committee (Plant Operations Review Committee - PORC) oversight. The management review included the generation of a status report. As of August 9, 1999, there were 20 open BMOs. About half of the open BMOs were less than one year old; however, one was from 1996 and four were from 1997. The licensee anticipated that five BMOs would remain open after completion of the Fall 1999 refueling outage. In comparison, 18 months earlier, i.e., just prior to the previous refueling outage, the licensee had 67 open BMOs. After the outage, eight BMOs remained open. The team determined that the decrease in open BMOs indicated improved attention to plant issues which included addressing degraded conditions and implementing the required equipment repairs and/or design changes.

c. Conclusion

The licensee made significant improvement in addressing degraded conditions and reducing the number of open BMOs. Also, the process for operability determinations (BMOs) was clarified.

E2.3 Engineering Backlog

a. Inspection Scope

The team reviewed the licensee's tracking of the engineering work backlog, including event reports (ERs), design changes, temporary modifications, and BMOs. The team also interviewed engineering management

b. Observations and Findings

Approximately two years ago, the licensee established an on-site engineering organization. The development of the new organization included bench marking of good engineering practices at other nuclear facilities. In conjunction with this review, in the fall of 1998 the licensee identified the need for tracking and trending the backlog of engineering work. The licensee defined backlog as the total work minus the planned work and their goal was to have all planned work completed such that the resultant backlog would be equal to zero.

From the time the backlog goal was made, engineering work progress had been reviewed by licensee management on a routine basis; but trend data had not yet been completely developed. At the time of the inspection, the event reports and commitment work-off rate were the only items being actually trended, i.e., they were being assessed on whether the quantity of items was increasing or decreasing relative to a goal. Other engineering activities were tracked by progress towards completion. Major work items were classified by priority in accordance with procedure AP0051, "SMRC [Station Modification Review Committee] Review and Approval Process." The team determined that on August 2, 1999, there were: 31 "A1" (mandatory) items, 47 "A" (high priority) items, 8 "B" (medium priority) items, and 10 "C" (low priority) items.

Trending of commitments included the total number of items and the quantity of items greater than 2 years old. Both groups had a declining trend. The licensee's goal for the commitments was to have less than 40 items greater than 2 yrs old by November 1999, and to reduce the total quantity of items to less than 600 by May 2000. The licensee tracked ERs against a goal of no more that 90 open. Earlier in 1999, the total open ERs oscillated around 75; since May 1999, the number had steadily increased to about 100, with most of the ERs being outage related. The information was summarized on a monthly basis in the Engineering Department SCAN Report.

c. Conclusion

The licensee was acceptably managing the engineering work backlog and was making good progress in developing trend data.

E3 Engineering Procedures and Documentation

E3.1 Design Modification Process

a. Inspection Scope (37550, 37001)

The team reviewed selected portions of the procedures for the development and safety evaluation of design changes to evaluate the adequacy of the guidance available to the engineering organizations for the design, installation, testing, and evaluation of plant modifications.

b. Observations and Findings

The current guidelines and requirements for the preparation, review, and approval of design modifications at VY were included in procedures AP 6008, "Vermont Yankee Design Change," Revision 0, and AP 0020, "Control of Temporary and Minor Modifications," Revision 21. Additionally, procedure AP 6002, "Preparing 50.59 Evaluations," Revision 7, delineated the guidelines and requirements for preparation, review, and approval of the 10 CFR 50.59 safety evaluations. To strengthen the guidance provided to the personnel responsible for plant configuration changes and improve the results of such changes, the licensee had recently developed Procedure AP 6008 and revised the other procedures.

The team's review of selected portions of the above procedures found them to provide sufficient guidance for the development and evaluation of design change packages, both temporary and permanent, and for the installation and testing of the plant modifications. The team also found the instructions to be sufficiently clear and the required references to other procedures and documents appropriately listed.

c. Conclusions

The procedures for the design, evaluation, installation and testing of permanent and temporary plant modifications were acceptable and sufficiently detailed to ensure proper implementation of the change process and to ensure that the change did involve an unreviewed safety question.

E5 Engineering Staff Training and Qualification

E5.1 System Engineering Training and Qualification

a. Inspection Scope (37550)

The team conducted interviews of system engineering and supervisory personnel to evaluate the ability of the system engineering group to support the plant organizations.

b. Observations and Findings

The system engineering organization has been in existence for approximately two years. The team's discussions with system engineers and their supervisors indicated various levels of experience among the system engineers. The organization was composed by a mix of experienced engineers drafted from other site organizations, experienced engineers obtained from other plants and industries, and engineers that were new to both VY and the nuclear industry.

Training of new system engineers was not structured. Typically, an engineer was assigned one or several systems and training was provided on an ongoing basis. Some engineers indicated that they had received general system training. All engineers were scheduled to participate in the training provided to the operators for the systems assigned to their responsibility.

As a result of the above and a high turnover, the effectiveness of system engineering appeared limited. Nonetheless, the engineers interviewed showed a strong desire to learn and willingness to assume the responsibility for their respective system performance. Similarly, management showed high interest in the system engineers taking on their respective system responsibility and to prevent problems. Communications with other engineering organizations was good and no concerns were identified by the team regarding the involvement of system engineering in the activities related to their systems.

c. Conclusions

The system engineering organization was relatively new and included several engineers that had either limited system experience or were new to the site. Therefore, the effectiveness of the organization appeared limited. Nonetheless, the engineers interviewed indicated a strong desire to take on the assigned responsibilities and be directly involved in the activities related to their respective systems.

E7 Quality Assurance in Engineering Activities

E7.1 Review of Self-Assessment Activities

a. Inspection Scope (37550)

The team reviewed the results of self-assessments conducted by engineering personnel and of the engineering program evaluations performed by the Quality Assurance (QA) organization to assess the effectiveness of the licensee’s self-assessment programs related to engineering activities. The team also conducted interviews of responsible engineering and QA personnel.

b. Observations and Findings

The QA organization has assigned one person who is solely responsible for the oversight and auditing of the engineering programs and activities. The team reviewed reports of QA audits performed during the past year. The reports were of good quality and findings indicative of a good audit program. The audits addressed the performance of the engineering organizations as well as specific engineering activities. The team determined that the findings had been clearly stated and that the recommendations for resolving the identified issues were appropriate. Discussions with QA personnel indicated that while findings were entered into the corrective action program and tracked to resolution, observations were not. The observations were, nonetheless, discussed with the proper discipline manager and an appropriate resolution was agreed upon. The observations as well as the findings became the basis for future audit plans. Audit results were informally trended.

In engineering, self-assessments were originally conducted without the benefit of self-assessment guidelines or procedures. A procedure was recently developed by one engineering group and the guidelines contained therein were used to develop procedures for the other groups. The team reviewed approximately fifteen engineering self-assessment reports completed between 1997 and 1999. The team found the reports to be of acceptable quality and to include negative as well as positive observations. The team also found that, although the depth of reviews varied with the organization or individuals, in all cases the reports included assessment of the specific program processes.

As in the case of the QA audits, recommendations were considered to be opportunities for improvement. Therefore, it was not always clear which recommendation had been accepted by the responsible organization and how it was being tracked for implementation. The licensee's recent revision of the design change process was partially the result of engineering self-assessments.

c. Conclusions

The engineering program evaluations conducted by the Quality Assurance (QA) organization and self-assessments conducted by Engineering were effective in identifying engineering program areas for improvement. The QA audits were broad in scope and addressed most engineering programs. Findings were formally tracked and observations appropriately reviewed and resolved.

E8 Miscellaneous Engineering Issues

E8.1 Motor-Operated Valve Position Indicators

a. Inspection Scope

The team noted that some motor-operated valves (MOVs) used a thin metal rod for local position indication. The team addressed the impact of the rod on the operation of the valve.

b. Observations and Findings

During a tour of the plant, the team observed that some of the emergency core cooling MOVs had a thin metal rod attached to the valve stem that extended outside the valve stem housing. The rod was used for local position indication. The team asked the system engineer whether the MOVs would be able to operate if the position indication rod was accidentally bent or if its outward movement became obstructed in some manner. The system engineer indicated that the rod was fabricated from a relatively soft metal as compared to the valve stem and housing. However, the licensee had not previously considered this scenario and no analysis had been performed to address the NRC concerns. The team will review this issue during a future inspection. **(IFI 50-271/99-11-02)**

c. Conclusion

No analysis was available to address the impact on the operability of some valves if the movement of the thin rod attached to the valve stem for local position indication was obstructed in some manner. This item will be reviewed during a future inspection.

E8.2 (Closed) LER 98-22-01: Control Valve Fast Closure and Turbine Valve Closure Scram Bypass Did Not Reset When Greater Than 30% Core Thermal Power Exceeded Due to Ambiguous Licensing Basis

The technical details associated with the subject Licensee Event Report (LER) were reviewed in NRC Inspection Report 98-10. The review resulted in a minor violation. The supplement to the LER provided a detailed root cause analysis of the event, as well additional long term corrective actions. The team considered the root cause analysis and the corrective actions to be acceptable. Based on the on-site review of the additional information contained therein, LER 98-22-01 is closed.

E8.3 (Closed) LER 98-26-00: Inadequate Equipment Control Results in Reduced HPCI [high pressure coolant injection] System Reliability

The technical details associated with this LER were reviewed in NRC Inspection Report 98-14. The description and analysis of the event, as contained in the LER, were consistent with the team's understanding of the event. Based on an on-site review of the licensee's analysis and associated corrective actions, LER 98-26-00 is closed.

E8.4 (Closed) URI 99-01-02: Design Control for RHR and CS Instrument Modification.

During the 1998 refueling outage, the licensee implemented a modification to upgrade the residual heat removal (RHR) and core spray (CS) system instruments to Regulatory Guide 1.97 Type "A" Category 1 criteria. An error during this modification resulted in the use of power supply cables that shared a common section of cable tray for the independent instrument channels. Due to this error, the applicable RG 1.97 separation criteria were not met. The NRC reviewed the operability determination prepared by the licensee previously and found it acceptable. The issue, nonetheless, was unresolved pending further NRC review of the adequacy of the design change and the licensee's efforts to evaluate the extent of condition.

To address the deficiency, on March 10, 1999, the licensee issued Minor Modification 99-005. The modification brought the design into compliance with RG 1.97. Additionally, as part of the closure of event report ER 99-0167 which had been opened to address the issue, the licensee conducted an apparent cause determination and an assessment of the potential existence of similar conditions. The licensee determined that the event was isolated and the result of incorrect assumptions in the design of the modification. Corrective actions, including training of engineering personnel regarding lessons learned, were completed in May 1999.

The team reviewed the licensee's evaluation of the event and the corrective actions and found them acceptable. Based on this review, the issue is closed. However, the licensee's failure to maintain the required separation between cables of redundant divisions is a violation of 10 CFR 50, Appendix B, Criterion III, Design Control. This non-repetitive licensee-identified and already corrected violation is being treated as a Non-Cited Violation, consistent with Appendix C of the NRC Enforcement Policy. **(NCV 50-271/99-11-03)**

E8.5 (Closed) URI 99-01-03: Design Control for RHRSW System Model.

The licensee identified that the design of the residual heat removal service water (RHRSW) pumps was not accurately reflected in the computer model used to verify the RHRSW system design in that the model did not include the net positive suction head (NPSH) loss due to the pump suction barrel. The NRC reviewed the subsequent LER and basis for maintaining operation (BMO) 99-44, but concluded that additional information was necessary regarding the development and verification of the RHRSW flow model and the consequences of the error to determine whether the model provided an adequate method for verifying the adequacy of the design.

During the current inspection, the team determined that, prior to using the computer flow model, the licensee had validated it through in-plant system testing and adjusted it to reflect the test results. Apparently, the licensee did not initially recognize that their failure to include the NPSH losses in the RHRSW model had contributed to the difference between the results predicted by the model and those obtained from the test. However, to compensate for the difference, the licensee had added a factor. Recently, when they determined the bases for the differences, they corrected the computer flow model and added a specific value for the NPSH losses due to the pump suction barrel.

Since adjustments were made by the licensee, following the original verification of the RHRSW flow model, the consequences for not specifically identifying the reason for the difference between the predicted and tested results were not significant. Also, because of the adjustments made to compensate for the differences between tested and predicted results, the model did provide an adequate method for verifying the adequacy of the design. Therefore, no violation of NRC requirements occurred. This issue is closed.

E8.6 (Closed) IFI 98-01-02: Potential degradation of RBCCW system containment isolation capability.

On February 20, 1998 the licensee reported a potential unanalyzed condition involving the susceptibility of the reactor building closed cooling water (RBCCW) system to a high energy line break (HELB). Specifically, the licensee postulated that the consequential failure of the RBCCW system following a HELB could compromise primary containment integrity. In their basis for maintaining operation, BMO No.98-05, the licensee evaluated the issue and concluded that the current RBCCW system was within the original design and licensing basis of the plant. Nonetheless, they maintained the associated event report, ER No. 98-0252, open pending further review. At the time of the event, the NRC reviewed the licensee's analysis and conclusion, but left the issue open pending determination by the licensee of needed corrective actions.

During the current inspection, the team evaluated the licensee's review of the issue and determined that the licensee had again concluded that the current system design was within the original design and licensing bases for the plant and that no additional actions were needed to resolve the issue. Concurrently, the NRC also reviewed the issue and similarly concluded that the system design was in accordance with the original VY design and licensing bases. This item is closed.

V. Management Meetings

X1 Exit Meeting Summary

The team presented the inspection results to members of licensee management at the conclusion of the inspection in a meeting, on August 27, 1999. The licensee did not indicate that any of the information presented at the exit was proprietary.

LIST OF ACRONYMS USED

BMO	Basis for Maintaining Operation
CFR	Code of Federal Regulations
CS	Core Spray
DBD	Design Basis Document
ER	Event Report
HELB	High Energy Line Break
IFI	Inspector Follow-up Item
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
NPSH	Net Positive Suction Head
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
PORC	Plant Operations Review Committee
QA	Quality Assurance
QC	Quality Control
RBCCW	Reactor Building Closed Cooling Water
RG	Regulatory Guide
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
SLC	Standby Liquid Control
SMRC	Station Modification Review Committee
TS	Technical Specifications

UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VY	Vermont Yankee

ITEMS OPENED, CLOSED, OR DISCUSSED

Opened

IFI 99-11-01	Surveillance Conducted without Entering a Limited Condition for Operation
IFI 99-11-02	Effect of Position Indicators on Motor-Operated Valve Operation
NCV 99-11-03	Inadequate design Control in the upgrade modification of RHR and CS RG 1.97 instruments

Closed

URI 99-01-02	Design Control for RHR and CS Instrument Modification
URI 99-01-03	Design Control of RHRSW System Model
IFI 98-01-02	Potential Degradation of RBCCW System Containment Isolation Capability
LER 98-022-00, 01	Control Valve Fast Closure and Turbine Valve Closure Scram Bypass Did Not Reset When Greater than 30% Core Thermal Power Exceeded Due To Ambiguous Licensing Basis
LER 98-026-00	Inadequate Equipment Control Results in Reduced HPCI System Reliability
NCV 99-11-03	Inadequate design Control in the upgrade modification of RHR and CS RG 1.97 instruments